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**Carolina Power & Light Company**
Harris Nuclear Plant
PO Box 165
New Hill NC 27562

MAR 3 1997

U.S. Nuclear Regulatory Commission
ATTN: NRC Document Control Desk
Washington, DC 20555

Serial: HNP-97-033
10CFR50.73

SHEARON HARRIS NUCLEAR POWER PLANT UNIT 1
DOCKET NO. 50-400
LICENSE NO. NPF-63
LICENSEE EVENT REPORT 97-001-00

Sir or Madam:

In accordance with Title 10 to the Code of Federal Regulations, the enclosed Licensee Event Report is submitted. This report describes a reactor trip which occurred during routine surveillance testing when a main feedwater isolation valve unexpectedly stroked shut.

Sincerely,

J. W. Donahue
Director of Site Operations
Harris Plant

MV

Enclosure

c: Mr. J. B. Brady (HNP Senior NRC Resident)
Mr. L. A. Reyes (NRC Regional Administrator, Region II)
Mr. N. B. Le (NRC - NRR Project Manager)

IE22

# CATEGORY 1

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:9703070191 DOC.DATE: 97/03/03 NOTARIZED: NO DOCKET #
FACIL:50-400 Shearon Harris Nuclear Power Plant, Unit 1, Carolina 05000400
AUTH.NAME AUTHOR AFFILIATION
VERRILLI,M. Carolina Power & Light Co.
DONAHUE,J.W. Carolina Power & Light Co.
RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 97-001-00:on 970131,automatic reactor tripped resulting from SG low-low level.Caused by inadvertent closure of MFIV (1FW-159) & faulty hydraulic relays.Shuttle valves replaced, SG level discrepancy investigated.W/970303 ltr.

DISTRIBUTION CODE: IE22T COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 3
TITLE: 50.73/50.9 Licensee Event Report (LER), Incident Rpt, etc.

NOTES:Application for permit renewal filed. 05000400

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NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/98



10 75

# LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

| FACILITY NAME (1) | DOCKET NUMBER (2) | PAGE (3) |
|---|---|---|
| Harris Nuclear Plant Unit-1 | 50-400 | 1 OF 2 |

TITLE (4)

Reactor Trip on low-low S/G level due to inadvertent closure of a Main Feedwater Isolation Valve (1FW-159).

| EVENT DATE (5) | | | LER NUMBER (6) | | | REPORT DATE (7) | | | OTHER FACILITIES INVOLVED (8) | |
|---|---|---|---|---|---|---|---|---|---|---|
| MONTH | DAY | YEAR | YEAR | SEQUENTIAL NUMBER | REVISION NUMBER | MONTH | DAY | YEAR | FACILITY NAME | DOCKET NUMBER |
| 01 | 31 | 97 | 97 | 001 | 0 | 3 | 3 | 97 | FACILITY NAME | DOCKET NUMBER 05000 |

OPERATING MODE (9): 1

POWER LEVEL (10): 97%

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| | 20.2201(b) | | 20.2203(a)(2)(v) | | 50.73(a)(2)(i) | | 50.73(a)(2)(viii) |
| | 20.2203(a)(1) | | 20.2203(a)(3)(i) | | 50.73(a)(2)(ii) | | 50.73(a)(2)(x) |
| | 20.2203(a)(2)(i) | | 20.2203(a)(3)(ii) | | 50.73(a)(2)(iii) | | 73.71 |
| | 20.2203(a)(2)(ii) | | 20.2203(a)(4) | X | 50.73(a)(2)(iv) | | OTHER |
| | 20.2203(a)(2)(iii) | | 50.36(c)(1) | | 50.73(a)(2)(v) | | Specify in Abstract below or in NRC Form 366A |
| | 20.2203(a)(2)(iv) | | 50.36(c)(2) | | 50.73(a)(2)(vii) | | |

LICENSEE CONTACT FOR THIS LER (12)

| NAME | TELEPHONE NUMBER (Include Area Code) |
|---|---|
| Michael Verrilli Sr. Analyst - Licensing | (919) 362-2303 |

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

| CAUSE | SYSTEM | COMPONENT | MANUFACTURER | REPORTABLE TO NPROS | CAUSE | SYSTEM | COMPONENT | MANUFACTURER | REPORTABLE TO NPROS |
|---|---|---|---|---|---|---|---|---|---|
| B | SJ | ISV | B350 | Y | | | | | |
| | | | | | | | | | |

| SUPPLEMENTAL REPORT EXPECTED (14) | | | EXPECTED SUBMISSION DATE (15) | MONTH | DAY | YEAR |
|---|---|---|---|---|---|---|
| YES (If yes, complete EXPECTED SUBMISSION DATE). | X | NO | | | | |

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On January 31, 1997, with the plant operating in Mode 1 at 97% power, an automatic reactor trip resulted from steam generator low-low level in the "A" Steam Generator (S/G). This trip occurred while performing quarterly surveillance testing on the Main Feedwater and Main Steam Isolation Valves (Operations Surveillance Test, OST-1018). During this testing, the Main Feedwater Isolation valve to the "A" S/G (1FW-159) was taken to the TEST position which should have caused the valve to stroke in the closed direction to the 90% open position. However, 1FW-159 stroked fully shut causing a decrease in feed flow and S/G level. The reactor operator performing the surveillance test in the main control room recognized that 1FW-159 had fully shut and attempted to restore feedwater flow the S/G by taking the control switch for 1FW-159 to the SHUT/RESET position and then holding the control switch in the OPEN position. The "A" feedwater regulating valve was also fully opened to increase feed flow. 1FW-159 would not re-open, thus resulting in a continued decrease in "A" S/G level. A locally stationed auxiliary operator reported that 1FW-159 opened approximately 2-3 inches then re-shut several times. Based on these conditions, the Unit Senior Control Operator commenced a load reduction to reduce steam demand and made plans to manually trip the reactor if S/G level decreased to 40%. (S/G low low level reactor trip setpoint is 38.5%) At 0438 hours, with lowest main control board indicated S/G level at approximately 44%, an automatic S/G low-low level reactor trip occurred. Due to the shrink in S/G levels, The Auxiliary Feedwater system started as required. All support systems functioned as required except for the pressurizer bank-A backup heater supply breaker, which tripped open after approximately ten minutes of operation. The plant was then stabilized in Mode-3 (Hot Standby).

This event was caused by faulty hydraulic relays (solenoid operated shuttle valves) that control the position of 1FW-159. Investigation revealed that the shuttle valve's o-ring seals were leaking by which prevented proper hydraulic operation.

Corrective actions included replacing the shuttle valves and satisfactorily testing 1FW-159, troubleshooting the pressurizer heater supply breaker and investigating the apparent steam generator level discrepancy.

NRC FORM 366A
(4-95)

U.S. NUCLEAR REGULATORY COMMISSION

# LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

| FACILITY NAME (1) | DOCKET | LER NUMBER (6) | | | PAGE (3) |
|---|---|---|---|---|---|
| | | YEAR | SEQUENTIAL NUMBER | REVISION NUMBER | |
| Shearon Harris Nuclear Plant - Unit #1 | 50-400 | 97 -- | 001 -- | 00 | 2 OF 2 |

TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

## EVENT DESCRIPTION:

On January 31, 1997, with the plant operating in Mode 1 at 97% power, an automatic reactor trip resulted from Steam Generator low-low level in the "A" Steam Generator (S/G). This trip occurred while performing quarterly surveillance testing on the Main Feedwater and Main Steam Isolation Valves (Operations Surveillance Test, OST-1018). During this testing, the Main Feedwater Isolation valve to the "A" S/G (1FW-159, EIIS Code:SJ-ISV) was taken to the TEST position which should have caused the valve to stroke in the closed direction to the 90% open position. However, 1FW-159 stroked fully shut causing a decrease in feed flow and S/G level. The reactor operator performing the surveillance test in the main control room recognized that 1FW-159 had fully shut and attempted to restore feedwater flow the S/G by taking the control switch for 1FW-159 to the SHUT/RESET position and then holding the control switch in the OPEN position. The "A" feedwater regulating valve was also fully opened to increase feed flow. 1FW-159 would not re-open, thus resulting in a continued decrease in "A" S/G level. A locally stationed auxiliary operator reported that 1FW-159 opened approximately 2-3 inches then re-shut several times. Based on these conditions, the Unit Senior Control Operator commenced a load reduction to reduce steam demand and made plans to manually trip the reactor if S/G level decreased to 40%. (S/G low low level reactor trip setpoint is 38.5%) At 0438 hours, with lowest main control board indicated S/G level at approximately 44%, an automatic S/G low-low level reactor trip occurred. The turbine tripped as expected on Turbine Trip/Reactor Trip P-4 and due to the shrink in S/G levels, both motor-driven Auxiliary Feedwater (AFW) pumps and the turbine-driven AFW pump started as required. All support systems functioned as required. The pressurizer bank-A backup heater supply breaker tripped open after approximately ten minutes of operation, but the remaining banks of pressurizer heaters restored pressurizer pressure to its normal band. The plant was then stabilized and maintained in Mode-3 (Hot Standby) to allow investigation into the reactor trip.

## CAUSE:

This event was caused by faulty hydraulic relays (solenoid operated shuttle valves) that control the position of 1FW-159. Investigation revealed that the shuttle valve's o-ring seals were leaking by which prevented proper hydraulic operation.

## SAFETY SIGNIFICANCE:

There were no adverse safety consequences associated with this event. With the exception of the pressurizer bank A backup heater supply breaker trip described above, all systems functioned as required to stabilize the reactor in Hot Standby.

This is being reported per 10CFR50.73.a.2.iv as an unplanned Reactor Protection System / Engineered Safety Feature actuation.

## PREVIOUS SIMILAR EVENTS:

There have been no previous reactor trips reported that were caused by faulty Main Feedwater Isolation valve solenoid operated shuttle valves.

## CORRECTIVE ACTIONS COMPLETED:

1. The solenoid operated shuttle valves for 1FW-159 were replaced and 1FW-159 was satisfactorily tested on February 1, 1997.
2. Troubleshooting was performed to determine the cause of the pressurizer bank-A backup heater supply breaker trip, but was inconclusive. A malfunctioning test switch was found and was replaced on February 1, 1997.
3. A review of archived plant process computer data was performed to investigate the apparent discrepancy between S/G levels last observed in the main control room and the low-low S/G level reactor trip setpoint of 38.5%. However, due to the computer data point 10 second update rate, an accurate level value at the time of the reactor trip could not be verified. The "A" S/G level channels were checked to determine if a calibration problem existed. No abnormalities were found and each of the channels were within the expected tolerance. These actions were completed on January 31, 1997.